NON-PUBLIC?: N
ACCESSION #: 8903150444
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Duane Arnold Energy Center (DAEC) PAGE: 1 OF 4

DOCKET NUMBER: 05000331

TITLE: Reactor Scram Due to Excessive Hydrogen Injection into Feedwater System During Preparation for a Special Test
EVENT DATE: 02/02/89 LER #: 89-003-00 REPORT DATE: 03/04/89

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Jeff S. Axline, Technical Support Engineer TELEPHONE: 319-851-7600

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE TO NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On February 2, 1989, with the plant operating at 100% power, during the preliminary steps of a hydrogen injection Special Test Procedure, a Reactor Scram occurred. The cause of this scram was high main steam line radiation levels caused by a larger than expected quantity of hydrogen being injected into the feedwater system. The root cause of this event was failure to re-verify test rig construction following replacement of its fittings. This resulted in incorrect placement of the flow orifices going unnoticed.

Long term corrective action to prevent recurrence of this type of event will be a revision of the Special Test Administrative Control Procedure. The revision will include the requirement for independent verification of test rigs and special valve lineups which will be used during special tests. This requirement will be applied to special tests, except those which have no potential for affecting safety-related systems or structures.

The reactor scram occurred as designed upon receipt of multiple main steam line high radiation signals. All rods inserted to the full-in position.

Throughout the event, vessel level and pressure were maintained within safe operating limits via proper response of feedwater level control and the safety relief valves as well as appropriate response by Operations personnel. This event had no effect on the safe operation of the plant.

END OF ABSTRACT

I. DESCRIPTION OF EVENT:

On February 2, 1989, with the plant operating at 100% power, during the preliminary steps of a hydrogen injection Special Test Procedure, a Reactor Scram occurred. This test (Special Test Procedure 156, In-Reactor Stress Corrosion Monitoring) involves the injection of various amounts of hydrogen into the feedwater system (EIIS System Code SJ) to determine the point at which stress corrosion cracking would be eliminated. To gain familiarity with the operation of the rig which would be used, preliminary test steps were performed on February 2. These preliminary steps consisted of bringing the test rig on line and injecting quantities of hydrogen in the range (6 SCFM) of that injected by the automatic hydrogen injection system used in normal operations. To prevent injecting hydrogen in excess of 6 SCFM, the automatic hydrogen injection system was to be valved out as the test injection system was valved in. The first step was reducing automatic hydrogen injection flow by 1 SCFM. The next step was to inject 1 SCFM of hydrogen via the test rig. The test rig's flow indicator was closely watched as the manual flow control valve was opened. While indicated flow was gradually being increased to 1 SCFM, steam line radiation levels began to increase as expected. Soon, however, steam line radiation levels began to reach alarm setpoints and were still increasing. Upon receiving steam line high radiation alarms, personnel at the test rig were immediately notified by the control room to shut the test rig down. The manual flow control valve was quickly closed but radiation levels continued to rise in the steam lines due to the delay in transport of hydrogen. Down stream piping of the test rig injection point was also isolated at this time. Shortly following this isolation, steam line radiation levels exceeded the scram trip point settings initiating an automatic scram.

II. CAUSE OF EVENT:

The cause of this scram was high main steam line radiation levels caused by a larger than expected quantity of hydrogen being injected into the feedwater system. Upon investigation of the cause of excessive hydrogen, it was determined that an incorrect flow orifice had been installed in the test rig injection line being used. The purpose of this flow orifice was to provide a differential pressure signal to the test rig's hydrogen flow indicator. The size of the orifice was such that the actual hydrogen injection rate was approximately fourteen times the amount indicated on the flow indicator.

As part of the original test rig construction, component placement was verified by the responsible test personnel and construction was determined to be satisfactory. During leak testing of the rig, various fitting leaks were found. Due to the leaks it was decided that another type of fitting would be used on the rig. While re-assembling the rig following fitting replacement, construction personnel interchanged the oxygen flow orifice with the hydrogen orifice. The flow orifices are identical in shape and size and were not identified as oxygen or hydrogen, therefore construction personnel had no reason to believe an error had been made. At this point leak testing was re-performed successfully but component placement on the rig was not re-verified.

The root cause of this event was failure of the responsible test personnel to re-verify test rig, construction following fitting replacement. This resulted in the incorrect placement of the flow orifices going unnoticed.

In addition to the factors which contributed to the interchanged flow orifices going undetected prior to the event, the following factors contributed to the error going undetected during operation of the rig. Although the operators in the control room were aware that an increase in steam line radiation would occur with an increase in hydrogen injection they were not provided with the expected quantitative increase in radiation level per increase in hydrogen injection. Also, the test participant operating the manual hydrogen injection valve did not have (available to him during the test) quantitative information on the amount of hydrogen flow per valve turn. Both of these additional factors may have caused the test to be aborted prior to the scram if they had been available during rig operation.

III. CORRECTIVE ACTION:

Short term corrective actions were to re-install the flow orifices in the correct locations and re-verify the entire rig construction. In addition, a steam line radiation level versus hydrogen flow graph and hydrogen flow versus turns of flow control valve graph was added to the test procedure. The test was successfully performed on the following weekend.

Long term corrective action to prevent recurrence of this type of event will be a revision of the Special Test Administrative Control Procedure. The revision will include the requirement for independent verification of test rigs and special valve lineups which will be used during special tests. This requirement will be applied to special tests, except those which have no potential for affecting safety-related systems or structures.

IV. ANALYSIS OF EVENT:

The reactor scram (EIIS System Code JC) occurred as designed upon receipt of multiple main steam line high radiation signals. All rods inserted to the

ull-in position. Coincident with the scram was the expected isolation of main steam isolation (Group I isolation) valves due to the steam line high radiation condition. Primary Containment Isolation Groups II-V (EIIS System Code JM) also functioned as expected upon vessel level falling below 170 inches. Approximately 10 seconds into the event HPCI (EIIS System Code BJ), RCIC (EIIS System Code BN), and LPCI (EIIS System Code BO) loop select were initiated on a LO-LO level signal. None of these safety systems injected as the LO-LO level signal only lasted approximately eight seconds and then cleared. RCIC was out of service at the time of the event and could not have automatically injected if the LO-LO signal had not cleared, however HPCI and LPCI were both available. Throughout the event, vessel level and pressure were maintained within safe operating limits via proper response of feedwater level control and the safety relief valves (EIIS System Code SB) as well as appropriate response by Operations personnel. This event had no effect on the safe operation of the plant.

IV. ADDITIONAL INFORMATION:

A review of the LER database did not indicate any similar events involving performance of Special Tests.

This event is being report pursuant to 10 CFR 50.73(a)(2)(iv).

ATTACHMENT 1 TO 8903150444 PAGE 1 OF 1

Iowa Electric Light and Power Company

March 6, 1989
DAEC-89-0155

Mr. A. Bert Davis
Regional Administrator
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Duane Arnold Energy Center
Docket No: 50-331

Op. License DPR-49
Licensee Event Report #89-003

Gentlemen:

In accordance with 10 CFR 50.73 please find attached a copy of the subject Licensee Event Report.

Very truly yours,

Rick L. Hannen
Plant Superintendent - Nuclear

RLH/JSA/go

cc: Director of Nuclear Reactor Regulation
Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D. C. 20555

NRC Resident Inspector - DAEC

File A-118a

Duane Arnold Energy Center 3277 DAEC Road Palo, Iowa 52324 319/851-7611

*** END OF DOCUMENT ***